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GEN-IV LFR Development: Status & Perspectives

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Abstract:

Since Lead-cooled Fast Reactors (LFR) have been conceptualized in the frame of Generation IV International Forum (GIF), great interest has focused on the development and testing of new technologies related to Heavy Liquid Metal (HLM) nuclear reactors. In this frame, ENEA developed one of the larger European experimental fleet of experimental facilities aiming at investigating HLM thermal-hydraulics, coolant chemistry control, corrosion behavior for structural materials, and at developing components, instrumentations and innovative systems, supported by experiments and numerical tools.

The present work aims at highlighting the capabilities and competencies developed by ENEA so far in the frame of the liquid metal technologies for GEN-IV LFR.

In particular, an overview on the ongoing R&D experimental program will be depicted considering the actual fleet of facilities: CIRCE, NACIE-UP, LIFUS5, LECOR and HELENA.

CIRCE (CIRColazione Eutettico) is the largest HLM pool facility presently in operation worldwide. Full scale component tests, thermal stratification studies, operational and accidental transients and integral tests for the nuclear safety and SGTR (Steam Generator Tube Rupture) events in a large pool system can be studied.

NACIE-UP (NAtural CIrculation Experiment-UPgraded) is a loop with a HLM primary and pressurized water secondary side and a 250 kW power Fuel Pin Simulator working in natural and mixed convection.

LIFUS5 (lithium for fusion) is a separated effect facility devoted to the HLM/Water interaction. HELENA (HEavy Liquid metal Experimental loop for advanced Nuclear Applications) is a pure lead loop with a mechanical pump for high flow rates experiments. LECOR (LEad CORrosion) is a corrosion loop facility with oxygen control system installed.

All the experiment actually ongoing on these facilities are described in the paper, depicting their role in the context of GEN-IV LFR development.

Keywords:

Gen-IV Lead cooled Fast Reactors, Lead-Technology, Research & Development, Experimental Infrastructures.

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1. Introduction

The LFR are considered the most promising technologies to meet the requirements introduced for GEN IV nuclear plants and they are being studied worldwide [USDOE & GIF, 2002], [USDOE & GIF,2014]. Their main characteristics are listed below:
Sustainability - the very low neutron absorption cross section and poor moderating power, allows to design fast-neutron spectrum with geometries characterized by a high coolant/fuel ratio and fuel bundle with high pitch to diameter ratio. The fast neutron spectrum and the breeding ration about 1 make possible an efficient utilization of excess neutrons and reduction of uranium consumptions with a reduction of the high radiotoxic waste thanks to a close fuel cycle.
Safety and Reliability - the coolant high molten point and the low vapor pressure allows a primary loop operating at atmospheric pressure and low temperatures; moreover, the high shielding capability against gamma radiation offers a great protection to the workers with very low doses. The good thermo-physical properties allow to design cores with a high pitch/diameter ratio with low pressure drops and consequently low power requested for pumping. In terms of passive safety, with an effective configuration it is possible to increase the system capability to remove the decay power in natural circulation regime with a consequent reduction of the active safety systems. The high density can avoid the risk of fuel compaction and subsequent achievement of critical conditions in case of core melting, promoting the dispersion phenomena, moreover, in case of breakage of the steam generator tubes, the high density of coolant reduces the risk of steam inlet inside the core. Finally, in case of loss of flow accident in the primary loop, the leaked lead will solidify in a very short time without significant chemical reactions, avoiding further loss of coolant and protecting the nearby structures and equipment.
Resistance to the Proliferation and Physical Protection - The MOX (Mixed Oxide Fuel) used contains actinides and it makes these systems unattractive for the extraction of weapon-usable materials. After all, the nuclear properties of the coolant can allow the realization of cores with a long life and not useful for the production of weapon-grade plutonium. The physical protection to the public and to the environment is assured by the coolant, which does not react with air and water at low pressure and reduces need for strong protection against the risk of catastrophic events deriving from natural causes or acts of sabotage, avoiding the chance of significant containment pressurization. Furthermore, the absence of inflammable substances reduces the risk of fire propagation.
Economy - The simple design reduces the building time, the capital cost and the operation and maintenance cost in order to offer a competitive price of the electricity generated. This is possible thanks to the favorable characteristics of the coolant chosen which allows the realization of low-pressure system with a steam generator integrated in the primary loop with a less complexity and dimension of systems. The absence of an intermediate loop makes possible thermal cycles characterized by a very high efficiency.
Research activities related to the lead and Lead-Bismuth Eutectic (LBE) technology development are ongoing in EU, with the design of two main systems: MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications), a subcritical research reactor using LBE as coolant and ALFRED (Advanced Lead Fast Reactor European Demonstrator). The extensive R&D (Research and Development) efforts undertaken are necessary in order to improve the knowledge and the experience performed in terms of design, operations, maintenance and materials for components. The LFR/ADS (Accelerator Driven System) technological issues concern the following main topics [P. Agostini et al., 2014].

- *Material studies and physical-chemistry coolant characterization* – it is necessary to assess the phenomena in which the lead and LBE are involved in LFR/ADS. The contamination of the lead by metal oxide and the corrosion of structural materials is the main issue in these systems. The long term exposure to liquid metal leads to embrittlement and degradation of structures and primary system components as vessel, internals, heat exchangers, fuel cladding.

- *Irradiation studies* – The activities focus on structural materials subject to fast neutron fluxes defining their resistance for thermal stresses and dpa (displacements per atom) and determining whether or not irradiation promotes embrittlement and corrosion attack by HLM. The main issues on the performance of materials are due to the corrosion on HML under irradiation, irradiation embrittlement, as well as neutron irradiation induced effects such as creep and swelling.
- *Thermal-hydraulic properties* – the relevant issues of the HLM thermal-hydraulics research are related to:
    - HLM pool thermal-hydraulics which identifies as main topics the study of forced convection flow (mixing, stratification, stagnant zones, surface level oscillations), natural convection flow (pressure drops, surface level oscillations), transition to buoyancy driven flow, fluid structure interaction and system response to seismic events.
    - Fuel Assembly thermal-hydraulics, with the main scope to define the assembly geometries to achieve optimal conditions for heat transfer between fuel rods and coolant in forced and natural circulation, also demonstrating the capability to maintain the geometrical features, withstanding to irradiation effects, high temperatures, mechanical loads and corrosion. Other issues which need investigation are: the sub-channel flow distribution, the cladding temperature profile and hot spot, the pressure drops, the vibrations induced by flow, the fluid structure interaction, consequence of a hypothetic core damage. The study of these topics is actually performed in ENEA (CIRCE, NACIE and HELENA facilities).

  The collection of experimental data aims to improve the knowledge of phenomena and processes, and to create a database for the validation of computer codes for the prediction of phenomena and processes relevant for design and safety.
- *Instrumentation* – the suitable instrumentation for LFR/ADS is an important challenge due to the high thermal loads, high temperatures, the corrosive environment, the fast neutron spectrum and the non-transparency of the coolant. The research activities aim to develop instrumentation capable to withstand the operating conditions of the (Heavy Liquid Metal Reactors) HLMRs and to maintain the reliability of the measurements for a long time.

The technological development of LFRs is summarized in *Figure 1*, according with the Technology Readiness Levels (TRL) approach.

	TRL	TRL Function	Generic Definition	Phase
achieved	1	Technology Down-Selection	•Basic principles definition	Screening
	2		•Technology concepts and applications definition	
Ongoing	3		•Demonstration of critical function •Proof of concept	Pre-qualification
	4	Final Process Selections & integration	•Lab-scale component validation	
	5		•Component validation in a relevant environment	Qualification
Further Development	6	Full-scale integrated testing	•System/subsystem model or prototype demonstration in relevant environment	
	7		•System prototype demonstration in prototypic environment	Demo
	8	Full-scale demo	•Actual system completed and qualified through test and demonstration	
	9		•Actual system proven through successful operations	

Figure 1: LFR Technology Development Overview according to the TRL approach

Here after are depicted the ongoing research activities, aiming at supporting the DEMO-LFR development, named ALFRED (Advanced Lead Fast Reactor European Demonstrator), presently ongoing by ENEA.

1. ALFRED Overview

ALFRED is conceived as the LFR demonstrator, having a core power of few hundreds of MWth which can be designed to be easily scaled up to the commercial size, and in parallel it aims at implementing prototypical technological solutions and components of a LFR First of a Kind (FOAK) with a target power of 250-300 MWe, which meet the Gen-IV sustainability goals.

At European level, an international consortium, FALCON, was established among Ansaldo Nucleare, ENEA and ICN in 2013 and joined by six supporting organizations in the following years. The consortium full-members, under the leadership of Ansaldo Nucleare, share the vision that ALFRED, supported by an adequate R&D program, is the only viable option to achieve the objective of a lead-cooled Small Modular Fast Reactor (SMFR) FOAK by 2035-2040 [M. Frignani et al., 2017].

The conceptual design of ALFRED (see *Figure 2*), in its role of European Technology Demonstrator Reactor (ETDR), was originally developed in the LEADER project (namely, Lead European Advanced Demonstrator Reactor), by leveraging on the experience gained through the collaborative research and innovation actions funded by the European Commission. In line with the strategic vision for the LFR deployment, and with the support of dedicated Italian, Romanian and Czech national programs, the ALFRED conceptual design is constantly improved, including a technical review oriented to preserve its safety performances [M. Frignani et al., 2017], while improving its readiness, flexibility and robustness.

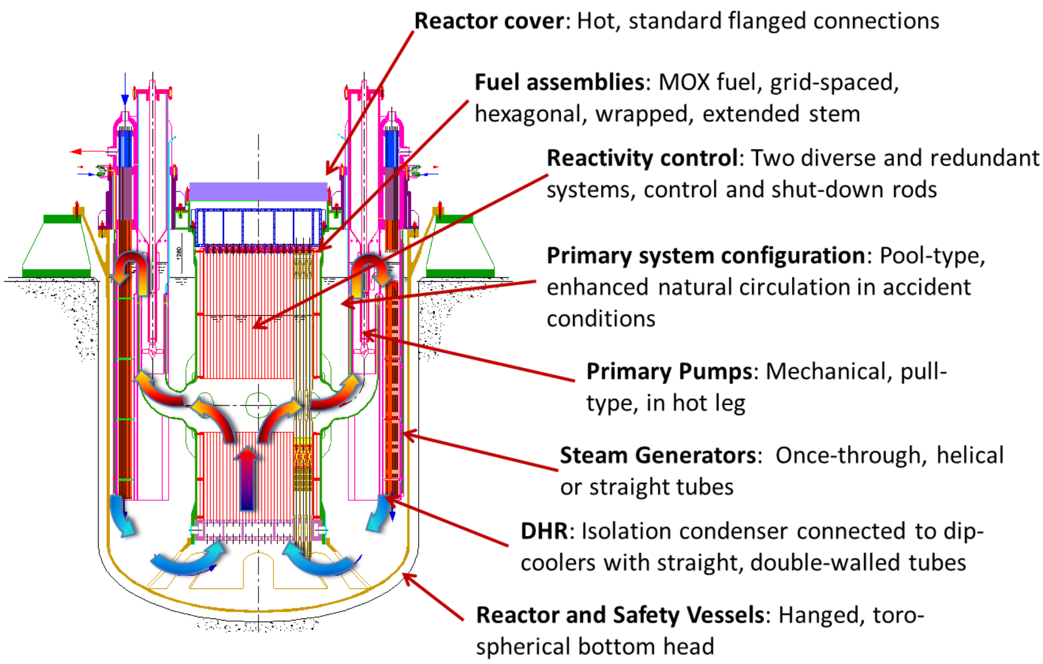


Figure 2: ALFRED Primary System Overview

The primary system pool configuration was revised in order to minimize thermal-hydraulic issues specific of liquid metal fast reactors (e.g., thermal stratification, thermal striping, free level fluctuations, gas entrainment) and their effects on immersed mechanical components.

The molten lead maximum temperature in the first stage of operation is limited to values fully compatible with selected materials, without the need for protection measures (like coatings or oxygen-driven self-passivation).
The core was provided with a hot channel to test improved fuel assemblies and, in particular, corrosion resistant coating (amorphous Al_2O_3 with nano-crystalline inclusions obtained by Pulsed Laser Deposition) of fuel pins, for future stages of operation at higher temperature.
No temporary in-vessel fuel storage of spent fuel assemblies is currently foreseen. Refueling operations will be carried out manipulating the fuel assemblies from the top, and keeping them under lead free-level to avoid any risk of loss of cooling for a stuck assembly during manipulation.
The lack of sufficiently mature under-lead in-service inspection techniques was by-passed, extending the concept of extractable components also to the core barrel and restraint systems. This will allow out-of-vessel inspection and repair, as well as testing of improved concepts for all the main primary components.
The main design features and parameters, outlined in *Table 1*, are in the final stage of optioneering studies.

Maximum core power (final stage)	300 MWth
Fuel assemblies	MOX fuel, grid-spaced, hexagonal, wrapped, extended stem
Reactivity control	Two diverse and redundant systems, control and shut-down rods
Primary system configuration	Pool-type, enhanced natural circulation in accident conditions
Reactor and Safety Vessels	Hanged, toro-spherical bottom head
Reactor cover	Hot, standard flanged connections
Steam Generators	Once-through, helical or straight tubes
Primary pumps	Mechanical, pull-type, in hot leg
Decay Heat Removal	Isolation condenser connected to dip-coolers with straight, double-walled tubes
Primary thermal cycle (1^{st} stage)	390-430°C
Materials (1^{st} stage)	Internal structures: AISI 316L Cladding: 15-15 Ti (AIM1)
Coolant chemistry control	Low oxygen content ($10^{-6} \div 10^{-8}$ wt.%)

Table 1: main candidate design features and parameters of the ALFRED reactor

2. Reactor Core Design

Among the R&D activities, the reactor core design plays a role of paramount of importance. The whole spectrum of core design activities is performed by ENEA, from concept definition to detailed neutronic, thermal/hydraulic and thermo-mechanic characterization. The holistic approach used is particularly effective for lead-cooled systems, to leverage on the potentialities, and to respect by-design the constraints, resulting from the inherent lead properties: in this way, the extreme flexibility of these systems can be fully exploited to magnify safety, sustainability and economics performances.

In particular, in this context ENEA provides for:

- Fuel Pin Design and Fuel Pin Mechanic Analysis
- Fuel Assembly & Fuel Pin Bundle Design (e.g. see *Figure **3***)
- Core Map Definition
- Control Rod & Safety Rod Conceptualization and Design
- Refueling Strategy & Burnup Performance
- Reactivity Feedback Coefficients Analysis
- Uncertainties propagation
- V&V (Verification and Validation) of neutronic codes

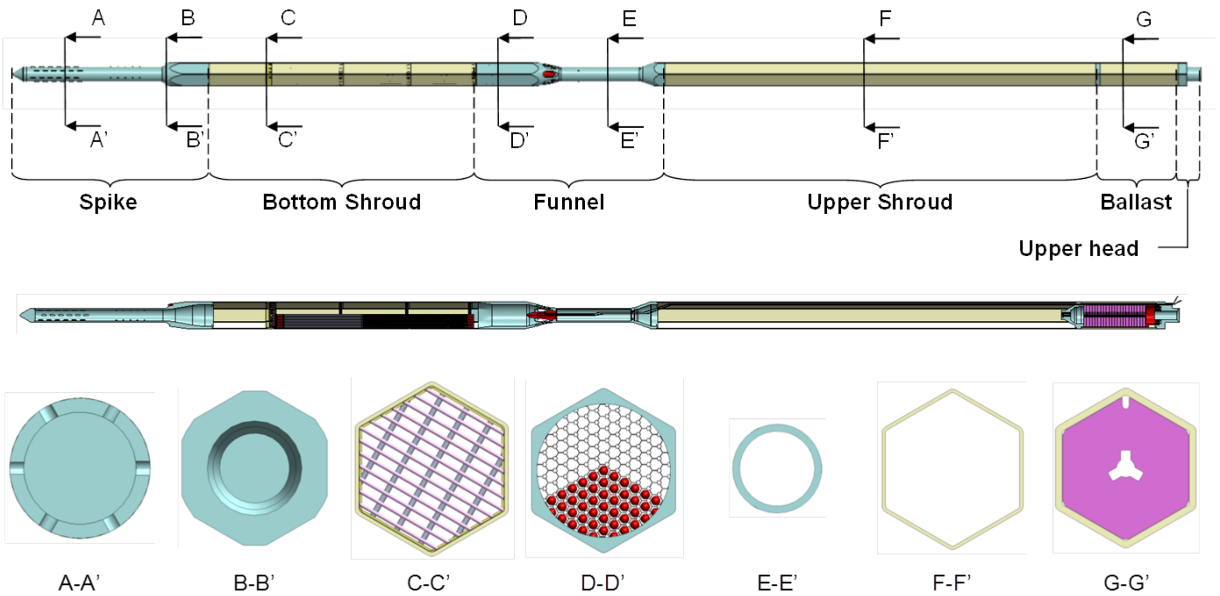


Figure 3: ALFRED Fuel Assembly Design

Figure 3 shows the ALFRED Fuel Assembly (FA) layout, as designed by ENEA. It is mainly composed by: an **inlet region,** named as "Spike", where several orifices are realized instead of a single large opening, to avoid the possibility of occlusions and flow blockage, a **wrapper section**, marked as "Bottom Shroud", consisting of an outer hexagonal cross-section enclosing the pins bundle and hosting the core active region, an **outlet region**, marked as "Funnel" characterized by a lower conical part with a set of openings realized to let the coolant flow outside the FA and a central cylindrical part, with reduced cross-section to realize a free volume above the core as upper coolant plenum, finally a **stem section**, named as "Upper Shroud" consisting of an hexagonal can extending above the lead free level, to allow the FA to emerge from the molten lead and engage the upper core plate.

3. Integral Tests

Among the experiments performed at ENEA, large relevance have integral tests performed in CIRCE facility at the Brasimone Research Centre. CIRCE is a pool facility consisting of a cylindrical vertical main vessel 8.5 m height filled with about 70 tons of molten LBE with argon as cover gas, LBE heating and cooling systems, auxiliary system for eutectic circulation and several test sections for the integral testing inserted inside the main vessel and fixed on the upper part through a coupling flange.

One of the last experiment performed on CIRCE deals with the ICE (integral circulation experiment) test section configuration used for the experimental campaign aiming at representing the transition from forced to natural circulation scenario in LFR [M. Tarantino et al., 2015]. The test section is represented in *Figure 4*, which consists of an electrical Fuel Pin Bundle Simulator (FPS) made up 37 heated pins disposed in a hexagonal wrapped lattice, with a nominal thermal power of about 800 kW. The LBE heated by the FPS flows inside the fitting volume and rises up to the separator (hot pool) passing through the riser. From the separator, the LBE is cooled in the Heat Exchanger and returns inside the main vessel (cold pool). The test section is also equipped with a DHR (decay heat removal system, 5-7% of nominal power) in order to reproduce the system behavior in case of an accidental scenario with loss of the heat exchanger (HX) heat sink and consequent reactor scram.

The CIRCE facility and the ICE test section are deeply instrumented with thermocouples located in several sections of the main components (FPS, RISER, HX, DHR, separator). In particular, *Figure 5* shows the LBE pool instrumentation installed in order to investigate on mixing and stratification phenomena. Several vertical rods have been installed into the pool allowing the TCs attachment at 17 different elevations for a total of 119 TCs. As shown in *Figure 5*, TCs on lines A, H and I allow measurements from the bottom side of the test section up to the FPS entrance, while TCs on lines B, C, D, E, F and G allow measurements up to 600 mm below the exit of the DHR. Furthermore, a Venturi Flow Meter is positioned in the “Feeding Conduit” (see *Figure 4*) to measure the LBE mass flow rate entering in the Fuel Pin Simulator.

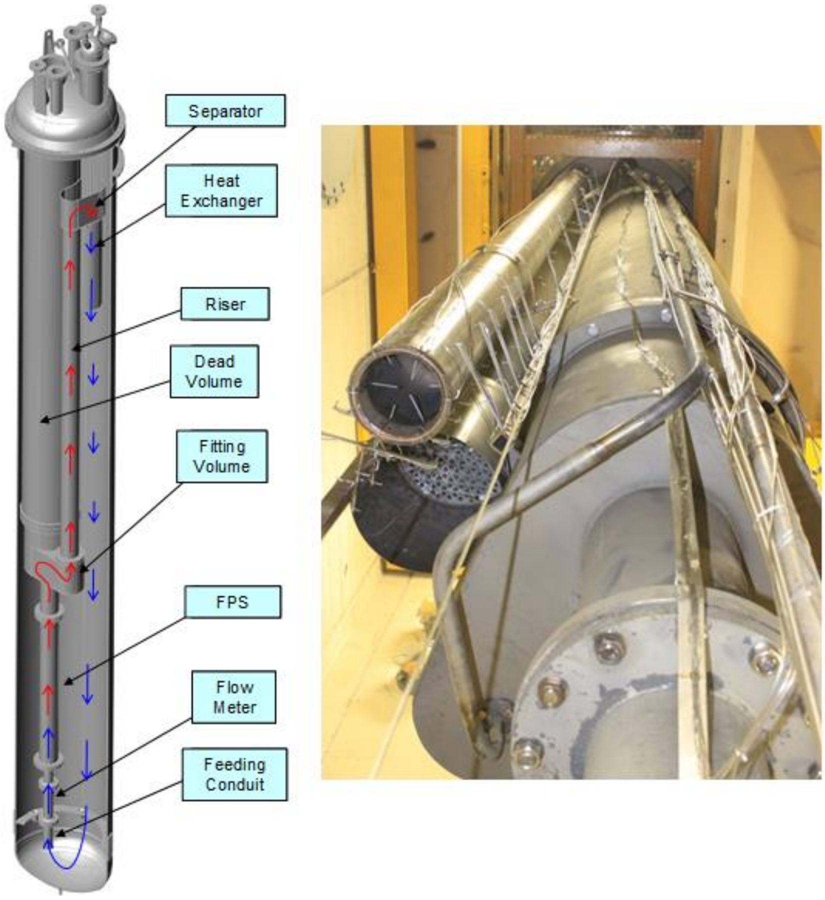
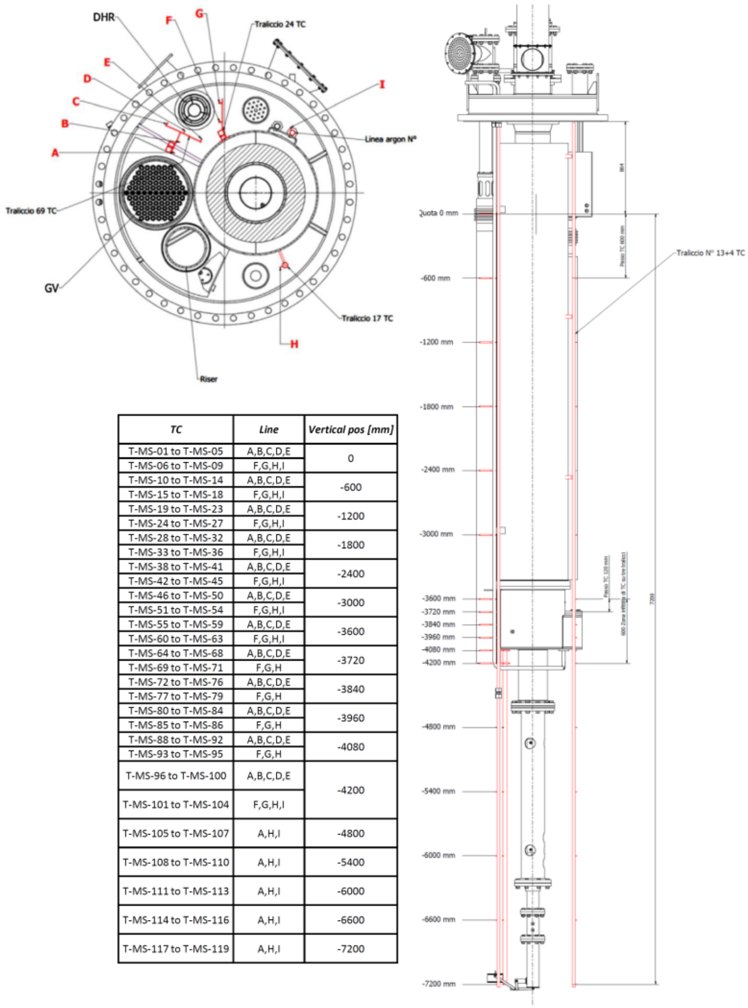


Figure 4: CIRCE configuration for integral tests campaign



TC	Line	Vertical pos [mm]
T-MS-01 to T-MS-05	A,B,C,D,E	0
T-MS-06 to T-MS-09	F,G,H,I	
T-MS-10 to T-MS-14	A,B,C,D,E	-600
T-MS-15 to T-MS-18	F,G,H,I	
T-MS-19 to T-MS-23	A,B,C,D,E	-1200
T-MS-24 to T-MS-27	F,G,H,I	
T-MS-28 to T-MS-32	A,B,C,D,E	-1800
T-MS-33 to T-MS-36	F,G,H,I	
T-MS-38 to T-MS-41	A,B,C,D,E	-2400
T-MS-42 to T-MS-45	F,G,H,I	
T-MS-46 to T-MS-50	A,B,C,D,E	-3000
T-MS-51 to T-MS-54	F,G,H,I	
T-MS-55 to T-MS-59	A,B,C,D,E	-3600
T-MS-60 to T-MS-63	F,G,H,I	
T-MS-64 to T-MS-68	A,B,C,D,E	-3720
T-MS-69 to T-MS-71	F,G,H	
T-MS-72 to T-MS-76	A,B,C,D,E	-3840
T-MS-77 to T-MS-79	F,G,H	
T-MS-80 to T-MS-84	A,B,C,D,E	-3960
T-MS-85 to T-MS-86	F,G,H	
T-MS-88 to T-MS-92	A,B,C,D,E	-4080
T-MS-93 to T-MS-95	F,G,H	
T-MS-96 to T-MS-100	A,B,C,D,E	-4200
T-MS-101 to T-MS-104	F,G,H,I	
T-MS-105 to T-MS-107	A,H,I	-4800
T-MS-108 to T-MS-110	A,H,I	-5400
T-MS-111 to T-MS-113	A,H,I	-6000
T-MS-114 to T-MS-116	A,H,I	-6600
T-MS-117 to T-MS-119	A,H,I	-7200

Figure 5: CIRCE instrumentation for Mixing and Stratification (M. Tarantino et al., 2015)

The purpose of these tests is to reproduce the thermal-hydraulic behavior of a HLM pool type reactor aiming to characterize the phenomena of mixed convection and stratification in a liquid metal pool in a safety relevant situation with transitions from nominal flow to natural circulation regime, reproducing a protected loss of heat sink (PLOHS) with a loss of flow (LOF) on the secondary side of HX, with a subsequent reactor scram and activation of DHR. The experimental activities proposed are in the framework of the HORIZON2020 SESAME (Simulations and Experiments for the Safety Assessment of MEtal cooled reactors) and MYRTE (MYRRHA Research and Transmutation Endeavour) Projects [SESAME Project, EC H2020], [MYRTE Project, EC H2020], with the purpose to support the qualification and validation of thermal-hydraulic numerical codes. The numerical simulation campaign has been realized in collaboration with University of Pisa and University of Rome "La Sapienza" (see V. Narcisi et al., 2017).

In this section, two tests have been presented. The main parameters are reported in *Table 2* while *Figure 6* and *Figure 7* show the main results (for a detailed description see M. Tarantino et al., 2015).

Test	Duration	Electrical Power	
		FC	NC
1	48 h	730 kW	50 kW
2	97 h	600 kW	23 kW

Table 2: Test matrix (M. Tarantino et al., 2015)

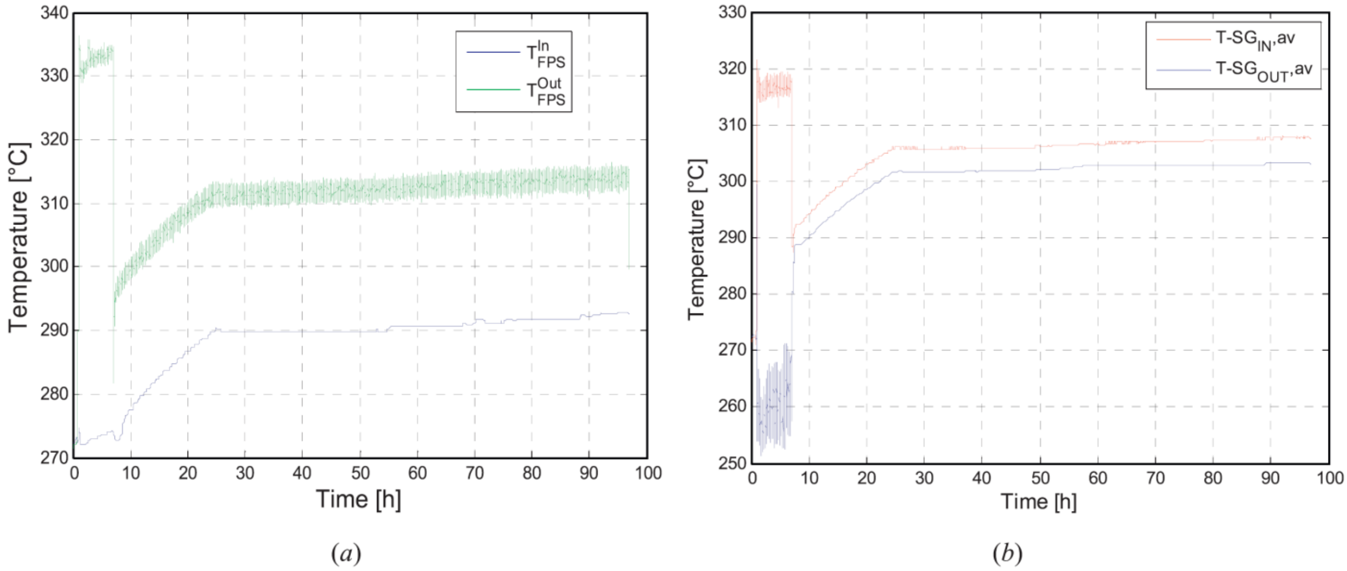


Figure 6: Test 1, average temperatures through the FPS (a) and the HX (b) (M. Tarantino et al., 2015)

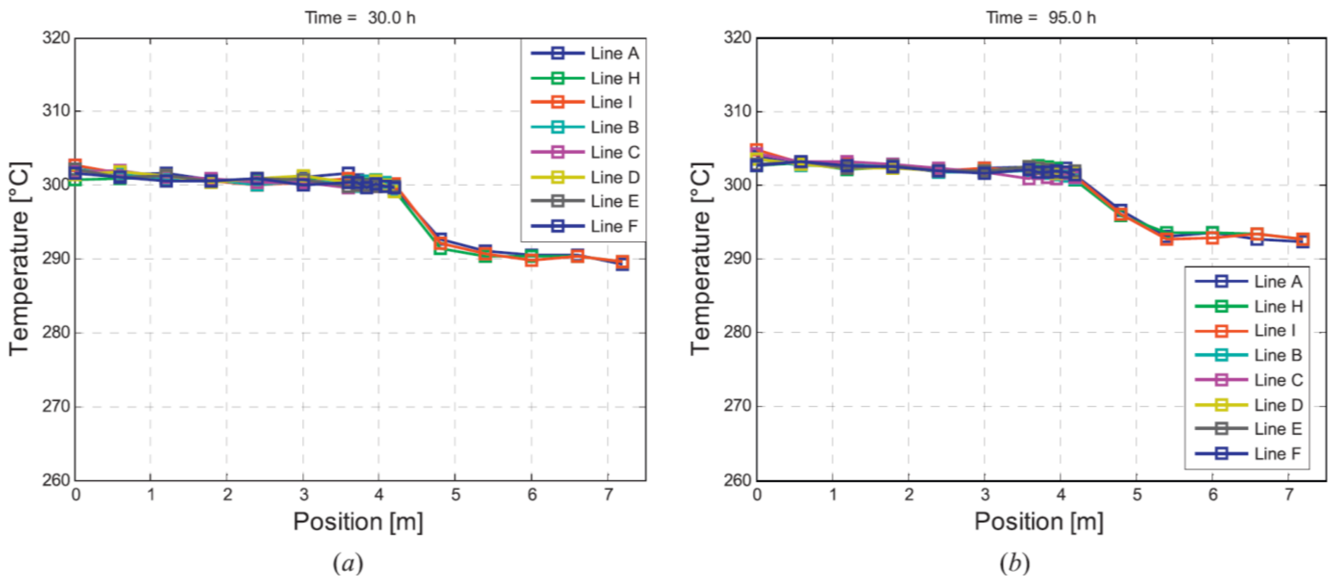


Figure 7: Test 2, temperatures inside the pool (M. Tarantino et al., 2015)

4. Fuel Pin Bundle Analysis

The NACIE-UP (NAtural CIrculation Experiment – UPgraded) experimental facility at the ENEA Brasimone Research Center allows to perform experimental campaigns in the field of thermal-hydraulics, fluid dynamics, chemistry control and corrosion protection, having as fundamental purpose to obtain experimental data on the specific topic of HLM cooled rod bundles, with the evaluation of the heat transfer coefficient in a wire-spaced fuel bundle cooled by LBE. The experimental campaigns performed are part of SEARCH FP7 EU project to support the development of the MYRRHA irradiation facility (SCK-CEN) [I. Di Piazza et al., 2016] and in the framework of the WP4 of the HORIZON2020 project SESAME which supports the development of European liquid metal cooled reactors (ASTRID, ALFRED, MYRRHA SEALER).

The NACIE-UP facility (*Figure 8*) consists in a rectangular loop with two vertical legs acting as riser and downcomer, an heat exchanger, an expansion tank and a wire-spaced 19-pin fuel bundle (*Figure 9*) with an active length of 600 mm for a total power of 235 kW, equipped with 67 thermocouples to monitor temperatures in different sub-channels and axial positions (*Figure 10*). The primary loop is equipped with an argon injection device, located inside the riser; it is composed of a 9 mm I.D. pipe, inserted from the coupling flange of the expansion vessel and ending with a hook-shaped pipe. The argon injection device allows to carry out transients from natural circulation to gas-enhanced circulation and vice-versa.

The experimental campaigns have been coupled with a numerical simulation activity with RELAP5-3D© thermal hydraulic system code in the frame of the development and validation of numerical codes for the HLMRs [SESAME Project, EC H2020]. The simulation activities have been realized in collaboration with other benchmark participants (i.e. UniPi, Uniroma1, GRS).

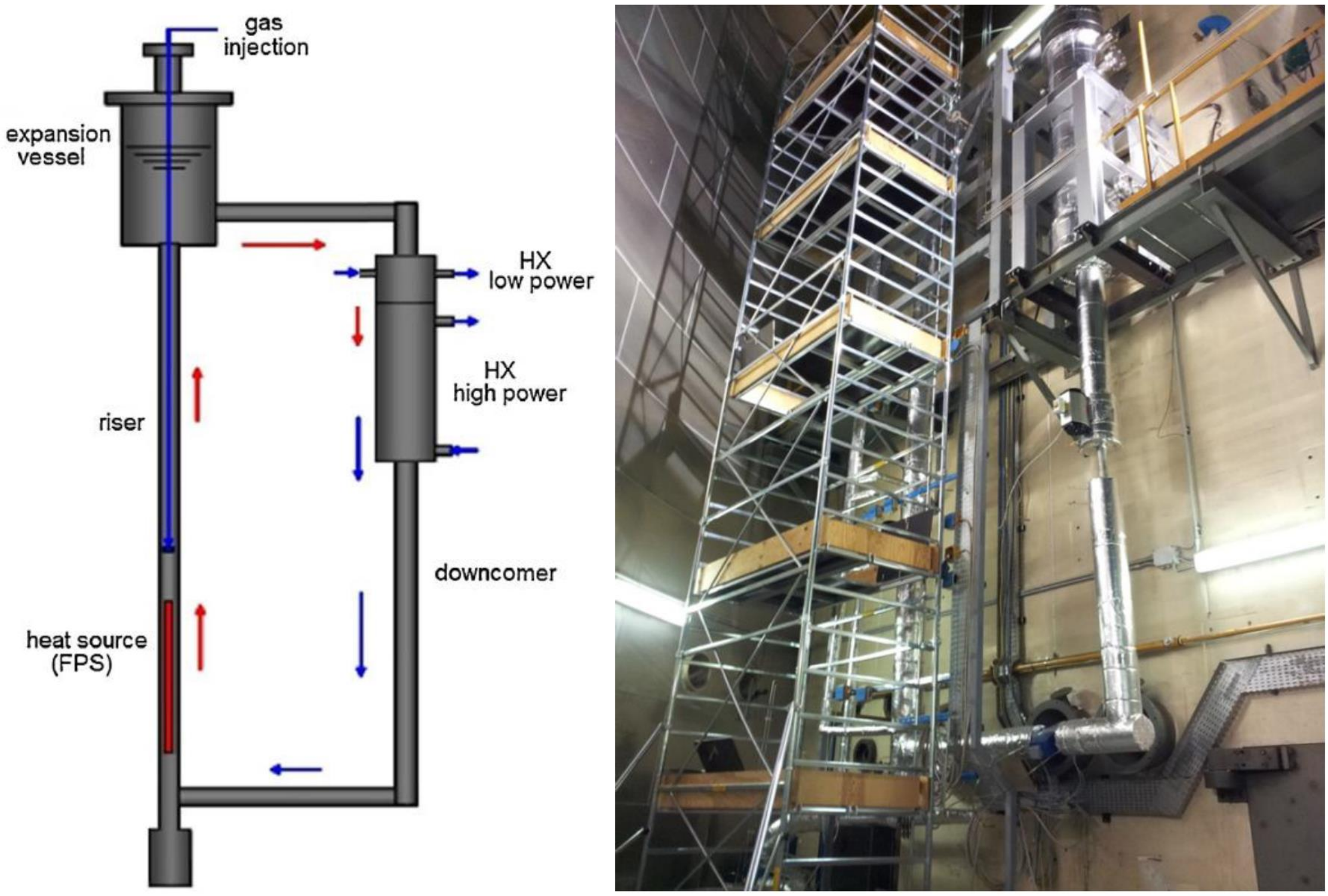


Figure 8: NACIE-UP facility

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Figure 9: Fuel Pin Bundle Simulator

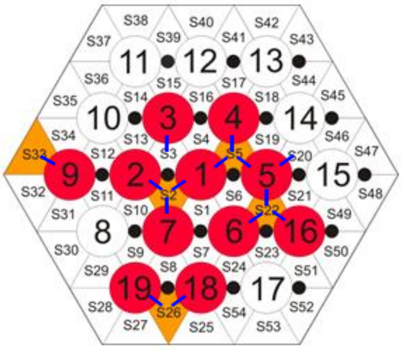

Figure 10: sketch of the monitored section of FPS (instrumented pins in red and instrumented sub-channels in orange, wall TC's in blue)

Two test cases are presented in *Table 3* while *Figure 11* reports the temperature distributions (for the complete test matrix and detailed results see I. Di Piazza et al., 2016).

Test	Q [kW]	Gas flow rate [Nl/min]	T_{in} [°C]	T_{out} [°C]	ΔT_{FPS} [°C]	$\dot{m}$ [kg/s]	u [m/s]
P43X0	43		226.1	404.2	178.1	1.61	0.248
P218	11	5.0	240.0	343.1	103.1	2.76	0.408

Table 3: Main parameters of tests P43X0 and P218 (I. Di Piazza et al., 2016)

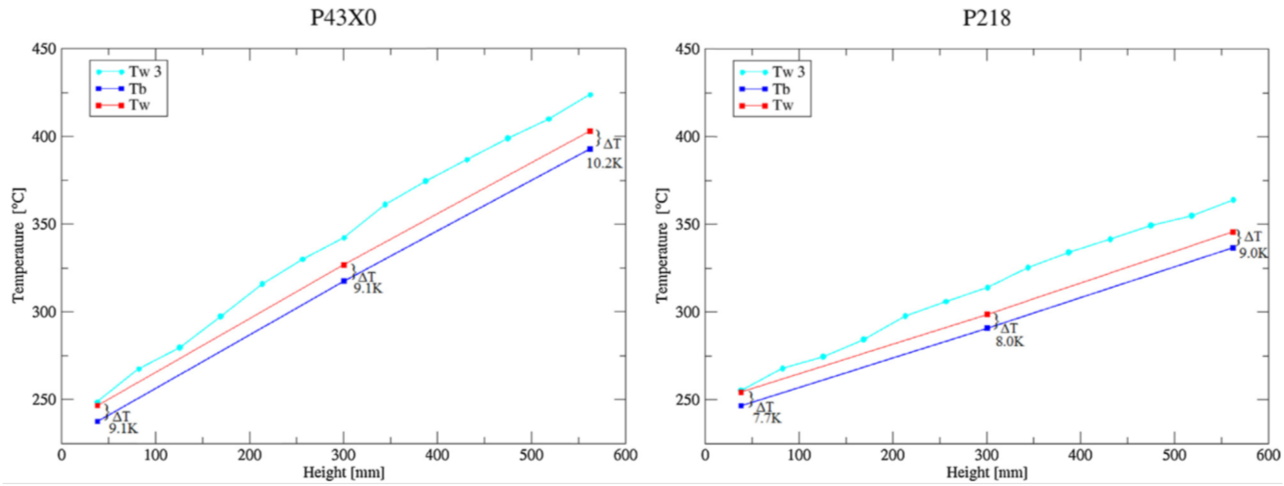

Figure 11: Test P43X0 and P218, axial distribution of average bulk and wall temperatures and pin 3 wall temperature (I. Di Piazza et al., 2016)

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5. Separate Effects Experiments

LFR and ADS designs are pool type reactors with the heat exchangers located inside the reactor vessel. This implies that the interaction between the secondary side coolant and the HLM may occur. An experimental campaign has been carried out about the safety analysis, investigating on the HLM-water interaction (i.e. PbLi, LBE, Pb) in a postulated Steam Generator Tube Rupture (SGTR) event in relevant configurations for HLMRs. The experimental tests were in the framework of the THINS (Thermal-Hydraulics of Innovative Nuclear Systems) project [THINS Project, EURATOM FP7] and have been carried out in the separate-effect facility LIFUS/Mod2, at ENEA Brasimone Research Centre. The main components of the facility, presented in *Figure 12* are: the interaction vessel (S1), the water injector tank (S2) and a dump tank (S3) which collects the vapour and HLM as consequence of interaction (disconnected for THINS tests).
The experimental campaign consists in two series of test with different water pressure, 40 and 16 bar. The experimental data collected comprise the pressurization time trends in the water injection tank, injection line and interaction vessel, and the temperature trends in the interaction vessel. A numerical post-test analysis of the first two tests of the low pressure series was carried out by the SIMMER-III code [A. Pesetti et al., 2016]. The main parameters of the two low pressure tests (T#5 and T#6) are reported in *Table 4*, while *Figure 13* and *Figure 14* show the comparison between the numerical results and the experimental data for T#5 and T#6 respectively.

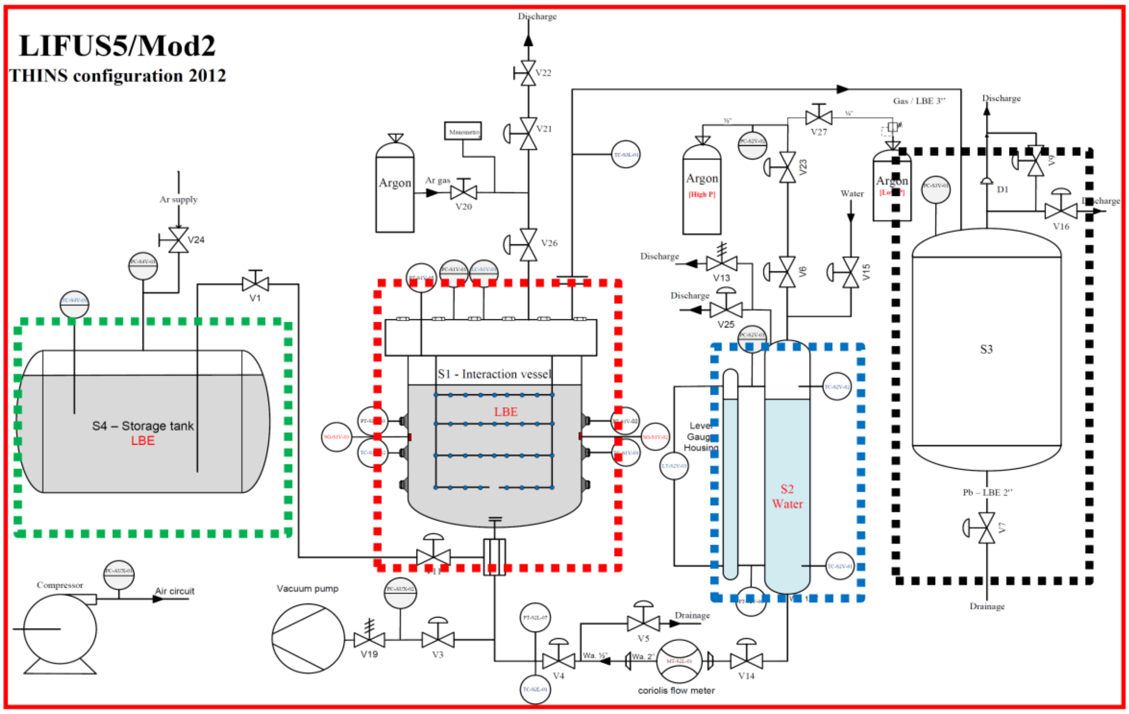

Figure 12: LIFUS5/Mod2 facility

Test	Interaction system	T_{LBE} [°C]	P_{H2O} [bar]	T_{H2O} [°C]	Argon/LBE volume [%]	Lasting time injection valve on [s]	Water injection penetration in S1 [mm]	Injector orifice diameter [mm]
T#5	S1	400	16	200	30	3	120	4
T#6	S1	400	16	180	30	3	120	4

Table 4: main parameters of tests #5 and #6 (A. Pesetti et al., 2016)

In particular, *Figure 13* reports the measured pressure time trend in the dome of the water tank (azure line), in the reaction vessel S1 (green line) and injection line and (grey line), while the calculated values for the reaction vessel S1 and the injection line are depicted red and blue, respectively. The SIMMER-III code predicts the pressurization of the injection line, in good agreement with the experimental data, with an overestimation during the transient phase and a good prediction of the experimental values achieved during the pressure plateau. *Figure 14* shows the mass flow rate measured by the Coriolis flow meter (red line) and computed by SIMMER-III code (blue line) and the water level in S2 (black line). In graph (a) of *Figure 14* the Coriolis and level meter notice the injection valve opening (occurring at t = 0 s) about 0.4 and 1 s later, respectively. The code, instead, shows an abrupt mass flow rate increase with an initial plateau of 1 kg/s and a decrease to 0 kg/s when the pressure in S1 equalize S2. In graph (b) of *Figure 14*, the level meter starts to measure the water injection more readily than in the test T#5, while the mass flow meter measures a delayed mass flow time trend. The SIMMER-III code computes a mass flow time trend similar to that of the previous test.

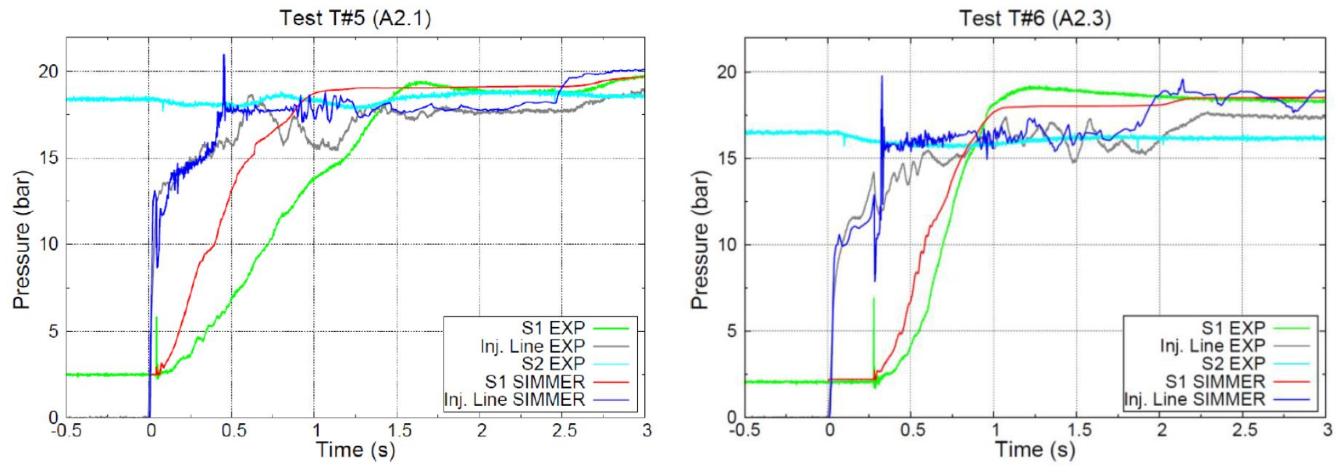

Figure 13: calculated and experimental pressure time trends of tests #5 and #6 (A. Pesetti et al., 2016)

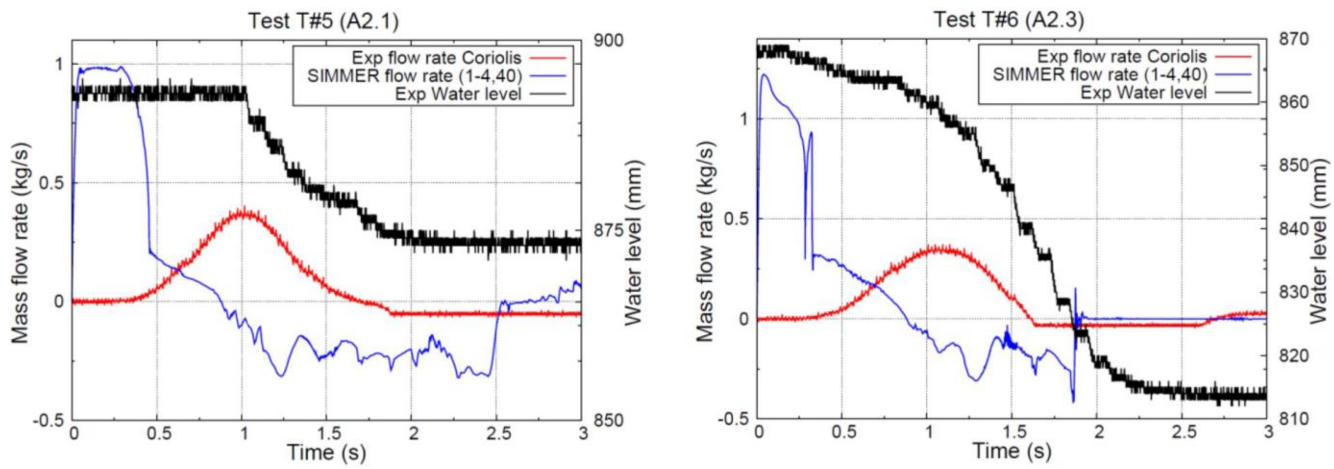

Figure 14: mass flow rate and level measurement of tests #5 and #6 (A. Pesetti et al., 2016)

6. Safety Analysis

The safety analysis is a key aspect for the GEN-IV reactors development, for this reason several research activities aims to assess all the most important and probable accidents which can occur in a LFR, evaluating the consequence for the integrity and the safety of the plant, and developing solutions for the plant protection. An important issues which has been investigated in this frame is the flow blockage in a fuel sub-assembly which leads to serious impacts on the safety of the plant, provoking the fuel assembly damaging or melting [R. Marinari et al., 2016]. In order to carry out the experimental campaign, the NACIE-UP experimental facility (described above) at the ENEA Brasimone Research Centre will be equipped with a new test section consisting in a Blocked Fuel Pin Simulator (BFPS), composed by 19 electrical pins arranged in an hexagonal wrapper for a total power of 250 kW. The BFPS will be used to simulate different flow blockage regimes, as shown in *Figure 15*, providing experimental data in support of the development of ALFRED. A preliminary CFD analysis of the BFPS experiment has been performed with ANSYS CFX 15 code and it has been presented in R. Marinari et al., 2016. *Figure 16* shows the clad temperature field in a blocked region of the Fuel Pin Simulator, obtained by CFD calculations.

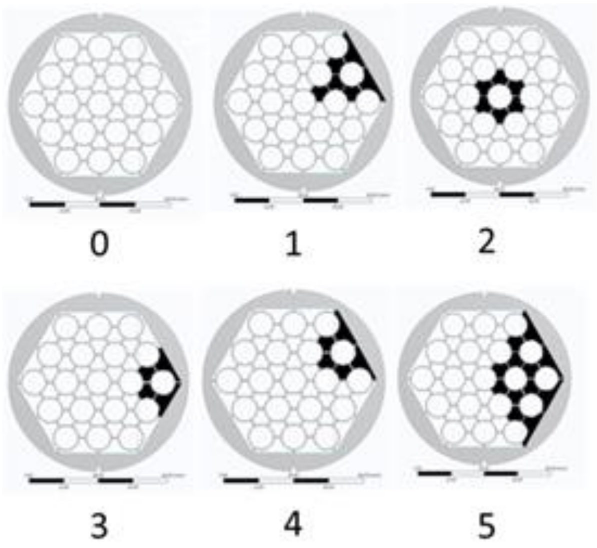

Figure 15: different blockage regimes(R. Marinari et al., 2016)

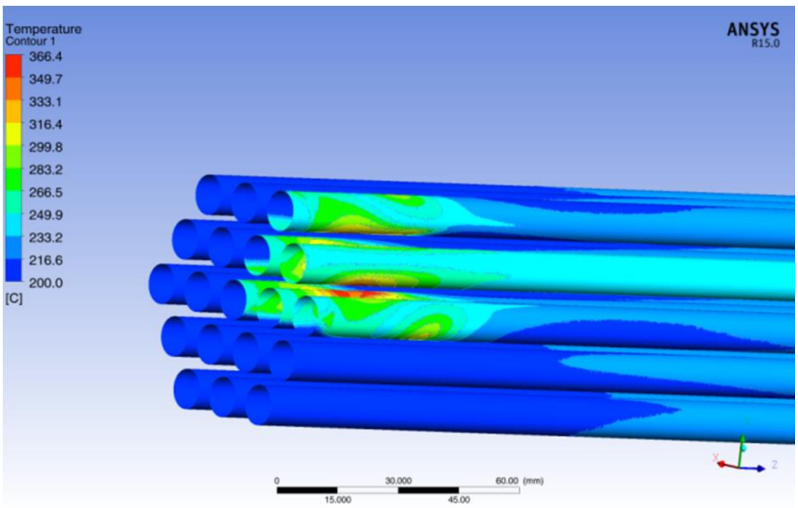

Figure 16: example of CFD calculation - clad temperature field in the region behind the blockage (type 1)(R. Marinari et al., 2016)

Another experimental campaign on the safety analysis has been realized in the framework of the MAXSIMA (Methodology, Analysis and Experiments for the "Safety In MYRRHA Assessment") project [MAXSIMA Project, EURATOM FP7], with the purpose to investigate on the Steam Generator Tube Rupture (SGTR) for the MHYRRA reactor. The experimental test section consists of four tube bundles, and each one reproduces a full scale portion of the tube bundle of the MHYRRA Primary Heat Exchanger (PHX). The test section has been implemented and tested in the CIRCE facility at the ENEA Brasimone Research Centre (for a detailed description see A. Pesetti et al., 2017). A top view of the test section installed in the CIRCE facility is reported in *Figure 17*. The experimental campaigns have been supported by a numerical simulation activity carried out with the 3D Cartesian SIMMER-IV code in order to conduce pre-test analysis before the execution of the experimental tests achieving knowledge about the system behavior during the test conditions. The Test Section consists in four SGTR-TSs, named as "A", "B", "C" and "D", for performing four SGTR runs. The conditions for each test are reported in *Table 5*. The graphs (a) and (b) reported in *Figure 18* show a comparison between the calculated (blue line) and experimental (red line) pressure time trends for the entire water-LBE interaction and first 10 s, respectively. The numerical results computed by SIMMER-IV code appear in agreement with measured data. After a good prediction of the first second of interaction, the code is able to compute the right timing of pressure oscillation at 1 and 2 s, with a low underestimation of pressure values between these instants. After the second pressure peak (t = 2 s) the computed pressure trend follows the measured one with a bit underestimation. Finally, after the pressure peak at ~7s, the computed pressure trend decreases following with a good agreement the experimental pressure trend.

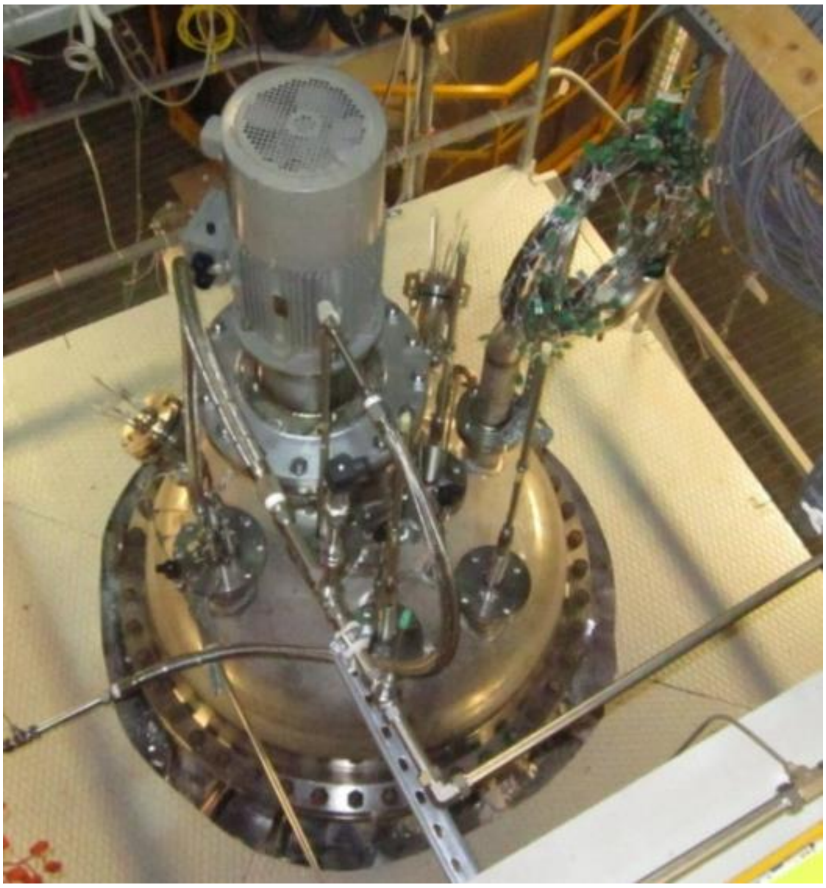

Figure 17: CIRCE Cover. MAXSIMA test section

BICs	SGTR-A	SGTR-C	SGTR-B	SGTR-D	
Interaction system	S100	S100	S100	S100	
S100 cover gas pressure [bar(a)]	1	1	1	1	
LBE temperature [°C]	350	350	350	350	
LBE mass flow rate [kg/s]	80	80	80	80	
Water pressure [bar(a)]	16	16	16	16	
Water temperature [°C]	200	200	200	200	
Water mass flow rate [g/s]	70	70	70	70	
Argon volume/LBE volume [%]	40	40	40	40	
First rank of tube pressure [bar(a)]	16	16	16	16	
Water Injection time in S100 [s]	3	3	3	3	
Isolation Valve	V9	V9	V9	V9	
Rupture position	MIDDLE	MIDDLE	BOTTOM	BOTTOM	
Injector orifice diameter [mm]	14	14	14	14	
Rupture disc 1 activation [bar(a)]	2.5	2.5	7	7	
Rupture disc 2 activation [bar(a)]	9	9	9	9	

Table 5: MAXSIMA Project test matrix (A: Pesetti et al., 2015)

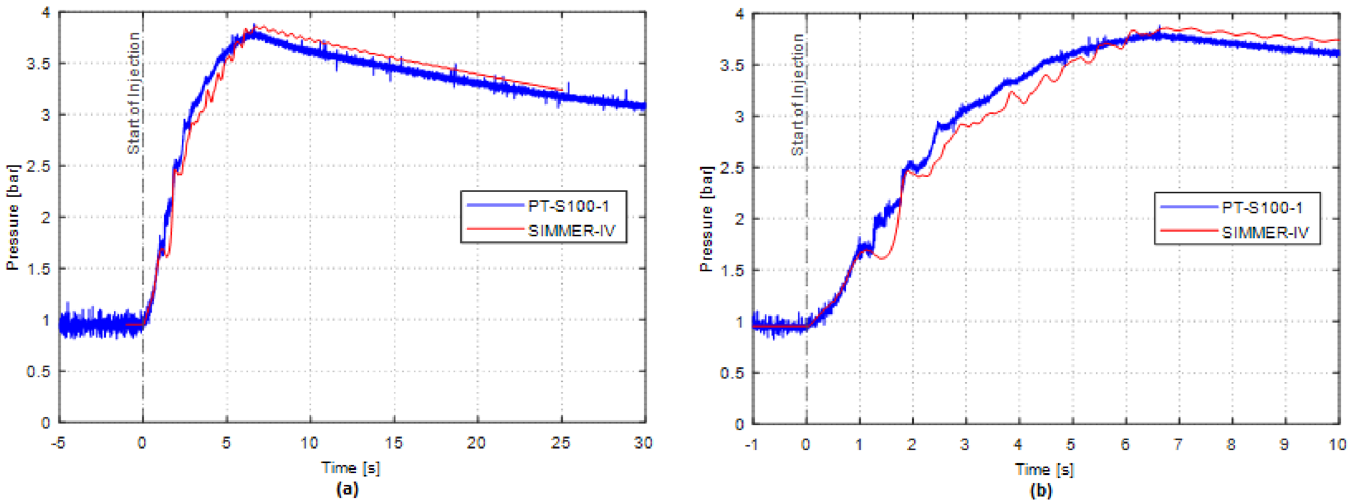

Figure 18: STGR-B - calculated and experimental pressure time trends in S100 cover gas((a) -5-30s), ((b) -1-10s)(A. Pesetti et al. 2015)

7. Coolant Chemistry and Structural Materials

The chemistry of liquid Pb is a key topic to be addressed for the development of LFRs. Two main issues have to be considered:

- HLM oxidation and coolant oxides deposition (mainly PbO) when oxygen is dissolved up to the solubility; such oxides affect HLM thermal-hydraulics, causing the plugging of the structures.
- Corrosion of steels; liquid Pb dissolves the alloy elements of stainless steels (Fe, Cr, Ni) and this is critical for structures exposed to the highest temperature (e.g. core).

Corrosion is influenced by oxygen dissolved in HLM and a sufficient level of oxygen is usually preferred and required to have the formation of a protective oxide layer (i.e. spinel oxide and magnetite) above steel surfaces.
According to this issues, CO in lead coolant have to be controlled in a proper range to avoid HLM oxidation and minimize steel corrosion via formation of magnetite above steel surface.
In general, the first requirement is mandatory for any nuclear system to avoid dangerous plugging. The second one is preferable but optional and its adoption depends on the strategy chosen for the specific system.
The strategy for ALFRED, the European DEMO-LFR [M. Frignani et al., 2015], [A. Alemberti et al., 2015], about coolant chemistry and oxygen control aims to work with an oxygen concentration (CO) in lead coolant much lower than the saturation at the minimum operating temperature (about 380°C, refuelling temperature). Specifically, the CO set for the operation is between 10^{-6} and 10^{-8} % wt. to reduce the chance of PbO formation even in case of local heterogeneities and potential deviations.
Figure 19 illustrates the oxygen concentration range, for a generic *Pb*-cooled system: the blue line corresponds to *Pb/PbO* equilibrium (maximum oxygen concentration limit), while full and dashed red lines corresponds to *Fe/*Fe_3O_4 equilibria (minimum oxygen concentration required for the formation of protective Fe_3O_4 above the steels) when a_{Fe}=1 and a_{Fe}<1 respectively. The optimal oxygen concentration is comprised between *Pb/PbO* and Fe_3O_4 equilibria. *Figure 19* also shows the maximum oxygen concentration range (dashed black box) and the preferred oxygen concentration (full black box) for ALFRED reactor. It can be noticed that the target CO for ALFRED is positioned further down than the range required to ensure both steel self-passivation and no HLM oxidation. This level of CO in principle guarantee the self-protection of the structures exposed to the lowest temperatures (e.g. the vessel at 400°C) but not the other structures exposed to higher temperatures. However, since passivation is not completely effective for steels exposed above 450-480°C because of the loss of protectiveness of the oxide scale, the use of other protection strategies are mandatory in Pb-cooled reactors even in presence of sufficient oxygen amount. In the case of ALFRED, the protection strategy consists in coatings, which have the advantage to protect the structures maintaining the mechanical and irradiation properties of the steel substrate (in particular, the swelling and the creep properties for fuel cladding tubes).

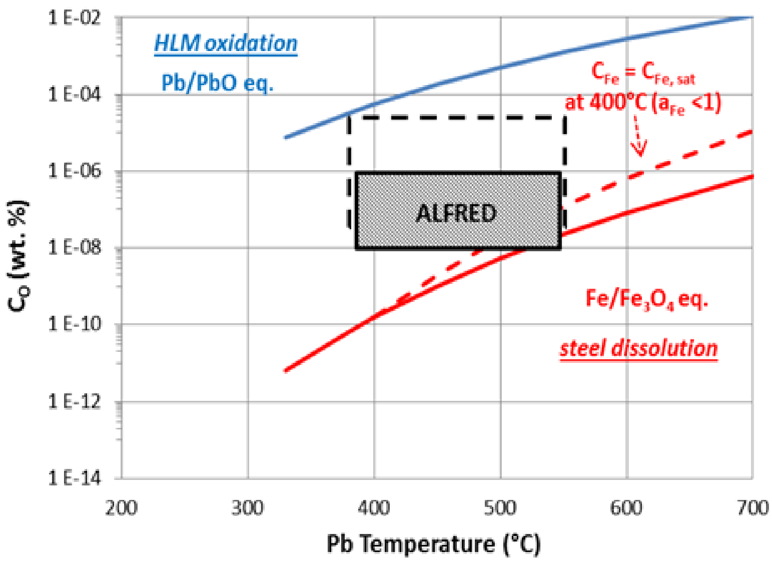

Figure 19. General CO range (dashed black box) and set CO (full black box) for ALFRED operation [S. Bassini, 2016].

The oxygen control at low content requires the adoption of procedures and on-line devices aimed to achieve and maintain the target concentration in the HLM, taking in account also potential deviations towards both higher CO (e.g. air contamination) and too much lower CO (e.g. oxygen adsorption by vessel walls for the passivation).

Oxygen sensors aimed to monitor the actual oxygen concentration in HLM are essential instruments. Concerning that, ceramic-based sensors have been developed in the last decade but their application mostly involves small scale systems (small vessels and loop facilities). Presently, ENEA is developing large oxygen sensors for the monitoring in pool system, where high-resistance is needed for the sensor to withstand the large pressure exercised by the large HLM volume in the vessel reactor.

Very important are the oxygen control procedures in start-up phases: the adoption of filtering, deoxygenating and degassing procedures before the filling of the main vessel with the HLM helps in having a high-purity liquid Pb and thus simplifying the oxygen control procedures into the reactor vessel [S. Bassini, 2016]. However, there is no experience in large pools about these operations and their feasibility has to be first assessed in large pool systems in order to define a suitable oxygen control system for ALFRED demonstrator.

For the effective control of the oxygen concentration in the main vessel, several techniques are considered. A good control of the oxygen concentration in HLM have been obtained in loop facilities and small vessels by using cover gas method with Ar-H2-H2O mixture, which is able to set a specific oxygen concentration in the liquid metal by using a given H2-H2O ratio in the cover gas. Other oxygen control methods considered are the injection of diluted H2 and O2 gases and PbO mass exchanger and oxygen getters. The use of H2 and O2 gas has the main advantage to allow rapid adjustment of the oxygen concentration [J-L. Courouau et al., 2004]. PbO mass exchanger is used as oxygen supply method and have been used and tested in several experiments [G. Muller et al., 2003], [P. N. Martynov et al., 2009]. Compared to O2 enriched gases, the device has the advantage to better control the release of oxygen (by controlling temperature and liquid metal flow rate) thus avoiding the risk of excessive oxidation of the liquid metal. Finally, oxygen getters (e.g. Mg) have been used in few experiments to reduce the oxygen concentration in liquid Pb alloy [C. Fazio et al., 2003] but they produce oxide powders that have to be entrapped to avoid coolant contamination.

About these techniques, a deep experimental campaign is ongoing in ENEA to assess the feasibility of these techniques in large HLM pools.

The low oxygen approach leads to an increase of the corrosive phenomenon and, furthermore, the corrosion effect is very critical for steel structures exposed to temperatures higher than 480°C even with a sufficient CO. This is because the oxide layer covering the conventional steels becomes at these temperatures permeable, thick and less protective. As a consequence, the use of protective coatings is mandatory to avoid severe damage of structures subjected to the highest temperatures in ALFRED (in particular, fuel cladding and fuel assemblies).

For fuel cladding tubes and fuel assembly structures, the cold worked 15-15Ti AIM-1 austenitic steel is considered the reference steel for ALFRED thanks to its good thermal creep and swelling properties under neutron irradiation.

The coating considered for these structure consists in amorphous Al2O3 with nano-crystalline inclusions obtained by PLD (Pulsed Laser Deposition). From the point of view of the mechanical behavior, the compatibility between steels and PLD-Al2O3 coating is remarkable: the nano-dispersed amorphous structure gives to the coating mechanical properties similar to those of the steel substrate and enhanced wear resistance [F. García Ferré et al. 2013]. Moreover, the compatibility of the coating with liquid lead is excellent up to 4000h at 550°C at low oxygen content [F. García Ferré et al. 2017]: no interaction is visible in static lead between lead and the coating (*Figure 20*). To the opposite, when 15-15Ti cold-worked is exposed to liquid lead in the same conditions, strong dissolution of the constitutive elements of the steel (in particular Ni and Cr) occurs [F. García Ferré et al. 2017].

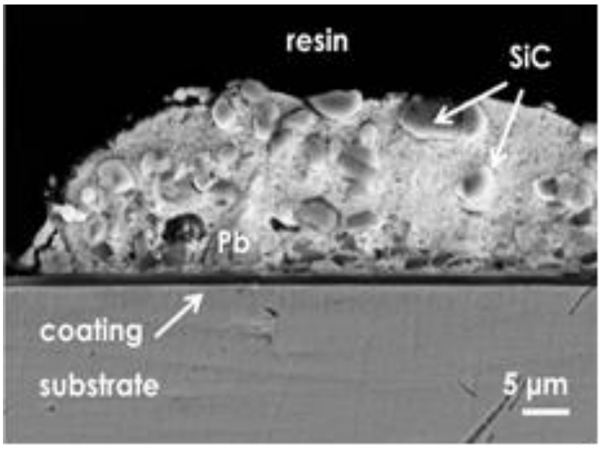

Figure 20. Coated 15-15Ti with PLD-Al2O3 after corrosion test in stagnant Pb, 550°C, 4000 h, 10-8 % wt. O. No interaction between Pb and the coating is visible.

AISI 316L/316LN is instead considered for reactor and inner vessel structures and primary pumps. For steam generator and DHR heat exchanger, the materials taken in account are AISI 316L or AFA steel. The latter has demonstrated to possess very good corrosion properties up to 550°C and even at low CO in HLM [J. Ejenstam et al., 2013], [J. Ejenstam, P. Szakálos, 2015], and its employment would not require the additional protection via coating. The extensive use of AFA steel also for core structures is not allowed since there are no data available about the behaviour under neutron irradiation. As a consequence, the use would be limited to the structures with negligible irradiation dose.

For the reactor vessel in AISI 316L/316LN, no coating is expected since self-protection is still guarantee at $CO=10^{-8}$ % wt. and T=380-430°C and corrosion effects can be considered acceptable and not dangerous for a large structure such as a vessel. A list of the main components of the ALFRED reactor with the relative selection of materials and coatings for low oxygen concentration is reported in *Figure 21*.

Materials and Coatings for low oxygen concentration ($C_O = 10^{-6} / 10^{-8}$ wt. %)

Component	*Min./Max Temp. Normal Operation (long term) (°C)*	*Max Temp. Accident Conditions (transient) (°C)*	*Max. Lead velocity (m/s)*	*Max. Radiation damage (dpa/y)*	*Max. Radiation damage (dpa)*	**Material**	**Coating**	**Notes**
Reactor Vessel	380÷430	500 (700 (1))	0,1	$< 10^{-5}$	0,0002 (40y)	**AISI316LN (ASTM)**	No	Back-up Liner of corrosion resistant steel (e.g. AFA)
Inner Vessel	380÷480	700 (1)	0,2	0,1	2,1 (20y)	**AISI316LN (ASTM)**	No Al diffusion coating by Pack Cementation	Back-up AFA steel
Diaphragm	380-480	700 (1)	0,2	0,1	(1,8) (20y)	**AISI316LN (ASTM)**	No	Al diffusion coating by Pack Cementation
Steam Generator	380÷480	700 (1)	0,6	$< 10^{-3}$	0,01 (20y)	**AISI316L (ASTM) 15-15Ti (DIN 1.4970)**	No Al diffusion coating by Pack Cementation	Backup: AFA steel; Alloy 800 coated
Fuel cladding	380÷550	750	2	20	100 (5y)	**15-15Ti (AIM1)**	Al_2O_3 by PLD	No buffer layer (direct coating deposition over the bulk material)
FA Structures	380÷500	700	2	19	100 (5y)	**15-15Ti (AIM1)**	Spacer grids & wrapper (outside): Al_2O_3 by PLD Wrapper (inside): overthickness	Back-up (for wrapper tube): Al_2O_3 by ALD or PLD (**to be developed in-tube deposition**)
DHR Heat Exchanger	380÷430	700 (1)	0,2	$< 10^{-3}$	0,01 (20y)	**AISI316L (ASTM) 15-15Ti (DIN 1.4970)**	No Al diffusion coating by Pack Cementation	Backup: AFA steel; Alloy 800 coated
Primary Pumps (impeller)	380÷480	700 (1)	15÷20	$< 10^{-3}$	0,01 (20y)	**AISI300 series**	Closed impeller: Al diffusion coating by Pack Cementation Open Impeller: Al_2O_3 by PLD	Backup (for open impeller ONLY): AlTiN coating by PVD

Figure 21. Materials and coatings selection for low oxygen concentration approach for ALFRED Reactor (DEMO-LFR).

Presently the approach depicted hereafter is under validation by ENEA, through a series of experimental campaigns planned on LECOR and HELENA loop (*Figure 22* and *Figure 23* respectively), both addressing structural material corrosion assessment and coolant chemistry demonstration.

Figure 22: LECOR facility

Figure 23: HELENA facility

8. Final Remarks

A part the R&D activities above described, ENEA is strongly involved in many others fields and sectors aiming at supporting the LFRs development.
In particular ENEA is providing efforts aiming at developing the following systems.

- Fuel manipulator (or handling) system. In current LFR design, refuelling and shuffling are performed remotely. The design and the operation of such machine, has to be tested before the installation in the reactor, for demonstrating its capability to fulfil his functions in reliable and safe way. This requires the assessment in an experimental facility of the prototype machine as well of its component for qualification purposes. In this frame CIRCE facility, might be in principle suitable for addressing the issues, thanks it high flexibility.
- Core Structures. Concerning the fuel channel assemblies and core support structure, mechanical and structural integrity of fuel assembly, in connection with fuel loading procedure, in a wide range of operating conditions need to be experimentally tested to verify the suitability of the design features, including flow induced vibrations and spacer grids fuel pin interactions. Furthermore, the bases for the core design needs to be investigated in connection with the influence of the neutronics on regulating and shutdown systems, which are designed to meet requirements set for the normal and abnormal (accident) conditions.
- Steam Generator. Apart the studies on safety above mentioned (mainly related to HLM-water interaction), the main qualification studies regard design validation, unit isolation on demand, pressure drop characteristics, component behavior in normal operation (e.g. forced, mixed and natural convection) and operational transients and accident conditions. Apart from the steam generator, in a reactor there are many several auxiliary systems which need to be qualified for nuclear applications. The following systems shall be considered: decay heat removal system, dip-coolers and isolation condenser, reactor vessel auxiliary cooling system (RVACS), fuel assembly transport system, spent fuel element transport and cooling system.

Concerning integral tests, they can provide data relevant for the full scale nuclear plant conditions, if the test facilities and the initial and boundary conditions of experiments are properly scaled (i.e. the scaling will not affect the evolution of physical processes important for the postulated accident scenario).On the other side, integral tests are fundamental for supporting the development and demonstrating the reliability of computer codes in simulating the behavior of a nuclear power plant, during a postulated accident scenario.

These considerations imply that integral tests are unavoidable and complex activities, which involve the following objective and areas of investigations:

- phenomena and processes of interest at system level and connected with design, safety and operation issues;
- simulations and analyses of a broad spectrum of accident scenarios;
- accident management procedures;
- component testing;
- scaling issue;
- generating databases for supporting licensing process;
- codes assessment and validation.

At present, only CIRCE facility is available and suitable to perform, with some extent, integral tests in Europe.

Concerning the fuel, ENEA (and many other stakeholders in LFR development) has non-manufacturing capability in this field. For this reason, in the short-term, an essential goal is to confirm that ready-to-use technical solutions exist (with uranium-plutonium MOX without minor actinides as the main candidate), so that fuel can be provided in timing with the LFR operation. In the long term, the potential for industrial deployment of advanced Minor Actinide bearing fuels and the possibility of using fuels that can withstand high temperatures to exploit the advantage of the high boiling point of lead will have to be investigated.

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